Analysis of the operator action and the single failure criteria in a SGTR sequence using best estimate assumptions with TRACE 5.0

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A B S T R A C T

Steam Generator Tube Rupture (SGTR) sequences in Pressurized Water Reactors are known to be one of the most demanding transients for the operating crew. SGTR are a special kind of transient as they could lead to radiological releases without core damage or containment failure, as they can constitute a direct path from the reactor coolant system to the environment.

The first methodology used to perform the Deterministic Safety Analysis (DSA) of a SGTR did not credit the operator action for the first 30 min of the transient, assuming that the operating crew was able to stop the primary to secondary leakage within that period of time. However, the different real SGTR accident cases happened in the USA and over the world demonstrated that the operators usually take more than 30 min to stop the leakage in actual sequences. Some methodologies were raised to overcome that fact, considering operator actions from the beginning of the transient, as it is done in Probabilistic Safety Analysis.

This paper presents the results of comparing different assumptions regarding the single failure criteria and the operator action taken from the most common methodologies included in the different Deterministic Safety Analysis. One single failure criteria that has not been analysed previously in the literature is proposed and analysed in this paper too. The comparison is done with a PWR Westinghouse three loop model in TRACE code (Almaraz NPP) with best estimate assumptions but including deterministic hypothesis such as single failure criteria or loss of offsite power. The behaviour of the reactor is quite diverse depending on the different assumptions made regarding the operator actions. On the other hand, although there are high conservatisms included in the hypothesis, as the single failure criteria, all the results are quite far from the regulatory limits. In addition, some improvements to the Emergency Operating Procedures to minimize the offsite release from the damaged SG in case of a SGTR are outlined taking into account the offsite dose sensitivity results.

1. Introduction

A Steam Generator Tube Rupture (SGTR) in a Pressurized Water Reactor (PWR) can lead to an atmospheric release bypassing the containment via the secondary system and exiting through the Power Operating Relief Valves (PORVs) of the affected Steam Generator (SG). That is the main reason why SGTR historically have been treated in a special way in the different Deterministic Safety Analysis (DSA), focusing on the radioactive release more than in the possible core damage, as it is done in the other Loss of Coolant Accidents (LOCAs).

In a SGTR event, the pressurizer pressure and level start to decrease at the same time that the Main Feedwater (MFW) of the faulted SG is decreasing to compensate the extra mass flow from the ruptured tube. In this phase of the transient, the operator can guess which SG is the faulted one with this decrease in the MFW or via radiation monitors (plant specific). The operators can start to manage the transient before the SCRAM, decreasing the turbine load and therefore the reactor power. They can also increase the charge flow and stop the letdown trying to compensate the break flow of the ruptured tube. Once the reactor is shutdown (manually o automatically), the operators have to identify the faulted SG and isolate it stopping the Auxiliary Feedwater (AFW) and closing the Main Steam Isolation Valve (MSIV) and the turbo-driven pump steam feed.

Once the faulted SG is isolated, the main goal is to stop the faulted tube leak via primary and secondary pressure equalization. Therefore, in a first step, the primary is cooled down via secondary depressurization through intact SG PORVs or steam dump. Then, the Reactor Coolant System (RCS) is depressurized to the faulted SG pressure via Pressurizer (PZR) spray or PZR PORVs. Later, the

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safety injection is stopped to prevent primary re-pressurization. If the primary and secondary pressures are diverging, a second depressurization is needed to stop the leakage. When the leakage is stopped, the primary circuit is slowly cooled and depressurized to Residual Heat Removal (RHR) conditions.

Therefore, the other aspect that makes SGTR different of the other LOCAs is the amount of human actions involved until the primary circuit is under RHR conditions. Several Emergency Operating Procedures (EOPs) are used in the accident management, Fig. 1. Once the reactor is tripped, the operating crew starts to read EOP E-0 "Reactor Trip or Safety Injection". Then, in steps 21, 25 or 29 (plant specific) the symptoms can lead to determine that there is an SGTR, going directly to EOP E-3 "Steam Generator Tube Rupture". The EOP E-3 finishes in EOPs ES-3.1, ES-3.2 or ES-3.3 which deals with long term cooling. If not, for example if the cooldown cannot be performed, there are other procedures that deal with the management called Emergency Contingency Action (ECA), like ECA-3.1 "SGTR with loss of reactor coolant subcooled recovery desired", Fig. 1.

Although it has been deeply study, the establishment of the hypothesis for the SGTR accident in the DSA is not an easy question, due to the complexity of the transient and the necessary operator actuation. At the beginning, the rule of the 30 min without operators action, developed for LOCA conditions, was applied to the SGTR DSA too, (Westinghouse, 1980). That hypothesis was based on the idea that the operators were capable of finishing the leakage of the ruptured tube within 30 min, so the most conservative assumption was to finish it in minute number 30.

Nevertheless, several real SGTR events demonstrated the difficulty of finishing the leakage within those 30 min. None of the operatońs crew involved in the SGTR events from 1975 to 1996 did it, and (MacDonald et al., 1996). The most relevant case both for the public and the industry was the SGTR happened in Ginna NPP in 1982, in which the leakage was finished 3 h after the accident started and the faulted SG released radioactive inventory to the atmosphere, although the dose was far away from the regulatory criteria of RG 1.195.

After the Ginna incident, the SGTR sequence attracted more attention than previously. The NRC was worried from the first SGTR accident to treat the different challenges observed in the real SGTR accidents was quick as it was reflected in different Unresolved Safety Issues (USIs), which last revision is (Fard, 2011) and its updated database (NRC, 2013):

- *USI 37*: Steam Generator Overfill and Combined Primary and Secondary Blowdown (Resolution date: 30/06/1985).

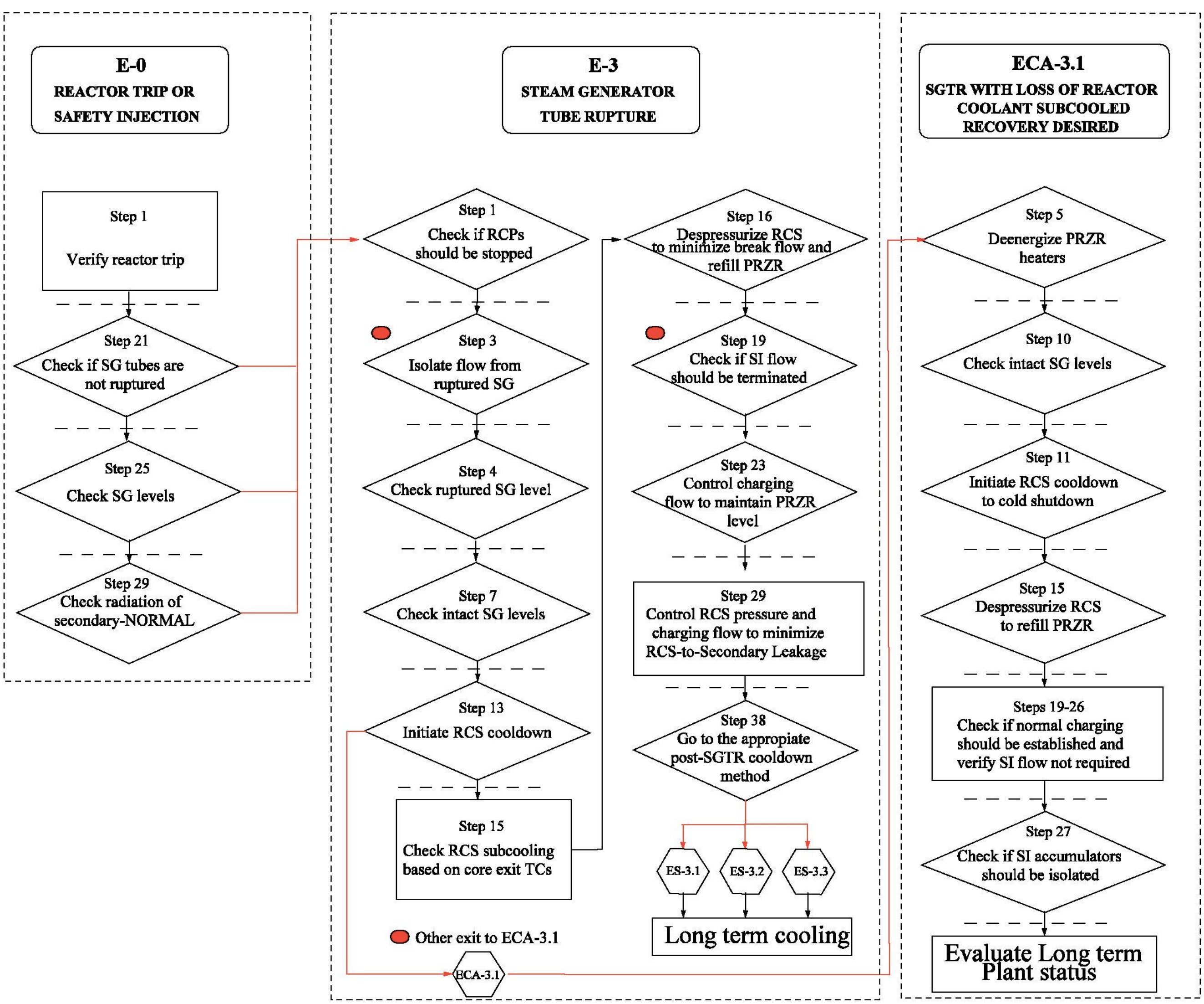

Fig. 1. Emergency operating procedures used in an SGTR accident management for a standard case.

- *USI 67.3.1*: Steam Generator Overfill (Resolution date: 30/06/1994)
- *USI 67.5.1*: Reassessment of SGTR Radiological Consequences (Resolution date: 30/06/1994).
- *USI 67.5.2*: Reevaluation of SGTR design basis (Resolution date: 30/06/1994).

In the USI 37 and 67.3.1, the SG Overfill possibility is faced. The main conclusions were that a safety grade SG level control system was needed to be implemented in all plants and together with operator specific training to avoid the SG overfill. Those recommendations were already included 2 years before Ginna NPP event in the Item I.C1 of the NUREG-0737 "Clarification of TMI Action Plan Requirements", (Eisenhut, 1980).

In the USI 67.5.1 it is stated that the "30 min rule" may be non-conservative and not consistent with operating experienceThe operating experience before the Ginna NPP event was reflected in the NUREG-0651 "Evaluation of steam generator tube rupture events", (Marsh, 1980). Therefore, as it said in the USI, implementation of this recommendation would allow the staff to upgrade SRP Section 15.6.3 and provide a better understanding and means to assess future SGTR events in operating plants relative to the consequence limits in RG 1.195. Nevertheless, the conclusion of USI 67.5.1 was that "Resolution of this issue was not expected to result in significant public risk reduction and, therefore, it was considered a LOW PRIORITY". However, as it was considered a licensing issue, it was resolved in the GI 67.5.1, "Reassessment of SGTR Radiological Consequences" of 1994,

(Gorman, 2000), that recommends substantial changes regarding the iodine partitioning, liquid carryover and iodine spiking hypothesis of SRP, but they were not implemented. It must be taken into account that the last revision of the SRP chapter 15.6.3 is from July of 1981, before Ginna accident.

In the USI 67.5.2 it is discussed the adequacy of changing the SGTR accident from Condition IV to Condition III, due to the high frequency of occurrence at that time (2E-2 accident/year), but it was closed without any action as it said "Had the frequencies of the conservative assumptions been included in a calculation of a design basis frequency, a much lower frequency would result", so there found no reason to change the SGTR to Condition III.

After Ginna incident, the NRC also started requiring in the Operating Licenses of the new plants that the owners demonstrate that the DSA case for SGTR was the most severe in terms of dose release, as it can be seen in the public correspondence with Catawba in 1986, (NRC, 1986) or with Diablo Canyon in 1987, (NRC, 1987). To face that requirement in an uniform way, a group of PWR owners (Shearon Harris, Byron and Braidwood, Catawba, Beaver, Valley Unit 2, South Texas, Millstone Unit 3, Ginna, Votgle, Watts Bar, Comanche Peak and Seabrook), (Ladieu, 1984), and Westinghouse Electric Company worked together to develop a methodology to analyze the SGTR accident taking into account the operating experience. The methodology was described in the WCAP-10698 "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill" (Lewis et al., 1987a) and its supplement "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident" (Lewis et al., 1987b). In the first document it is described how a SGTR analysis should be done to make sure that a faulted SG overfill cannot occur under the most conservative assumptions. The second document is used to perform an analysis to verify that the dose release under the most conservative assumptions is lower than the regulatory limit.

Both calculations, SG overfill and offsite dose, had different single failure criteria and use realistic operation time for the manual actions performed in the transient by the operators following the Emergency operating Procedures (EOPs). This methodology was not required by the NRC for the plants which have the operating license before the Ginna NPP incident in 1982, except Ginna NPP itself, as it can be seen in some public documents, see (Meyer, 2009) and (Schimmel, 2011). The selection of the single failure is plant based, and it has to be demonstrated that the chosen one is the most conservative. The other hypothesis are more general: Loss of Offsite Power (LOOP) is assumed in all of them and the initial conditions (temperature average, reactor power, etc.) are quite similar for all the cases found in the public literature.

The most conservative single failure criteria for SG overfill analysis is usually an AFW malfuntion like in Watts Bar (Monahan and Lewis, 1992) or the failure of the electrical bus that feeds the AFW, like in Beaver Valley (FENOC, 2006); or the failure of one intact SG PORV to open when demanded for the depresurization, like in Harris Nuclear Plant (Westinghouse, 2000) and Braidwood and Byron Plants (Borton, 2011a). For Point Beach (Nextera, 2010), Turkey Point (Kiley, 2011), Prairie Island (Schimmel, 2010) and Donald C. Cook (Stang, 2001) there is no single failure criteria as it is stated in their operating license for the DSA.

The most conservative single failure criteria for the offsite dose calculations is less variable, as the chosen one for all the cases available in the public literature is the failure of the faulted SG PORV to close after opening when the SG is isolated. The variability is indeed caused by the estimated time for each NPP to close locally that faulted valve, which is very plant specific: 11 min for Watts Bar (Monahan and Lewis, 1992), 10 min for Beaver Valley (FENOC, 2006), 20 min for Harris Nuclear Plant (Westinghouse, 2000) and 30 min for Byron and Braidwood plants (Borton, 2011a).

For the operating action time, each plant has their own values for the different manual actions performed from the operators, as it can be seen in Table 1. Some of the actions are not fixed, and the time spent in them is calculated by the thermal hydraulic code. In the case of Westinghouse calculations of SGTR the thermal hydraulic code used is LOFTTR2, (Monahan and Lewis, 1992).

The results of the SG overfill analysis show differences between the plants: some plants consider the margin to overfill the volume of the SG and the steam lines until the isolation valve. Other plants consider only the SG volume. For many of them the maximum level of the faulted SG is below the top of the upper dome, like in Watts Bar (Bochman, 1992), Harris Nuclear Plant (Westinghouse, 2000), Prairie Island (Schimmel, 2011), Byron and Brainwood (Borton, 2011b) and Turkey Point (Kiley, 2011). However, in the Beaver Valley analysis, the secondary side of the SG is completely filled and the steam line is partially filled with water, but the NPP owner considers that there is enough margin because the steam line is not filled until the isolation valve, Section 5.4.2.4 of (FENOC, 2006). In the case of the offsite dose analysis the results are much more similar and far away from the regulatory limit of RG 1.195.

There is at least other alternative to WCAP-10698 in the USA found in the public literature, the methodology developed by Kansas Gas & Electric Company (KGEC), owners of Wolf Creek and Callaway, the SNUPPS (Standardized Nuclear Unit Power Plant System) plants. In this methodology, WCAP-16265, (Young, 2004), there is no overfill calculation, however there are two different offsite dose calculations. The first one includes the failure of the faulted SG PORV within the SCRAM for 20 min. The second one includes the overfill of the faulted SG due to the AFW malfunction, assuming that the faulted SG PORV fails if there is liquid release.

European countries like Belgium, United Kingdom, Germany and France have developed alternative methodologies to the classical FSAR one, (Dutton, 1994):

- The failure of the faulted SG PORV at closing is contemplated in the Belgium, French and UK methodologies. This case is similar to those analysed in several USA NPPs.

Table 1
Different times for operator actions during a SGTR DSA (Westinghouse design).

Plant	Generic Times for W plants	Watts Bar	Beaver Valley	Harris NP	Prairie Island	Turkey Point	Braidwood/ Byron Stations
	PWR-3L	PWR-4L	PWR-3L	PWR-3L	PWR-2L	PWR-3L	PWR-4L
Reference	(Bochman, 1992) (pp. 31)	(Bochman, 1992) (pp. 31) (Monahan and Lewis, 1992) (pg 13)	(FENOC, 2006) (pp. 5–333)	(Westinghouse, 2000) (pp. 6–3-12)	(Schimmel, 2011) (pp. 34, enclosure 2)	(Kiley, 2011) (pp. 4, attachment 1)	(Borton, 2011b) (Table II-2, pp. II-13)
Operator action	Time (s) (AVG/MAX)	Time (s) (AVG/MAX)/ Used in the calculation	Time (s)	Time (s)	Time (s)	Time (s)	Time (s)
Identify and isolate ruptured SG	600/600	978/1080/900 or Calculated by LOFTTR2 at 30% NRL	900 from SCRAM	720 or Calculated by LOFTTR2 at 30% NRL	Calculated by LOFTTR2 at 35% NRL	1200	1080
Operation action time to initiate Cooldown	240/300	420/540/429	144	300	1140 from SCRAM	1680 from SCRAM	180
Cooldown of RCS			Calculated by TH code (LOFTTR2 for W calculations)				
Operator time to initiate depressurization	120/120	150/240/147	240	240	240	360	240
Depressurization			Calculated by TH code (LOFTTR2 for W calculations)				
Operator action time to initiate SI termination	60/60	132/192/244.2	180	180	120	180	180
SI termination and pressure equalization			Calculated by TH code (LOFTTR2 for W calculations)				

- The Main Steam Line Break (MSLB) coincident with the SGTR is an additional offsite dose calculation case in the French, UK and Germany methodologies.
- For the French case, the SGTR of one tube is cathegorised as a Condition III event, whereas the multiple SGTR or the MSLB with coincident SGTR are considered as Condition IV events for N4 plants, (Libmann, 1996). It implies that a double ended SGTR has to accomplish with Condition III dose limits.
- For the Belgian case in a similar way the SGTR was re-categorized in 1989, from Condition IV to Condition III due to its high frequency, (Parmentier and Delalleau, 2008). Some years after that re-categorization, due to the SG replacements and the inspection and maintenance improvements, the SGTR has changed to Condition IV again in 2006 as it is considered less frequent than before the improvements.

There are other methodologies, but for different kind of reactors, like the one done by AREVA for CE plants like St. Lucie, (Anderson, 2011). There is also another one done by AREVA for the Oukiloto 3 NPP EPR, as by Finnish regulatory guide YVL 1.0 "Pressure management during PWR primary-to secondary leaks shall be so arranged that no coolant discharges to the environment are required", than can be seen in (Israel, 2006). In the AP1000 SGTR analysis of Chapter 15 it is assumed that no operator action during the transient is necessary, (Westinghouse, 2007). For both the EPR and the AP1000, the single failure chosen for offsite dose calculation is that the faulted SG PORV stucks open, like in the WCAP-10698 methodology. Both EPR and AP1000 calculate the Margin to Overfill as an additional calculation too. For the APWR license in the USA, the same cases are considered as for the EPR and AP1000, (Mitsubishi, 2012).

Table 2
EAB and LPZ Accident Dose Criteria from RG 1.195 (NRC, 2003).

SGTR case	Dose criteria (Sv)		
	Whole body (Sv)	Thyroid (Sv)	Analysis release duration
Fuel damage or pre-accident spike	0.25	3	Affected SG: time to isolate; unaffected SG(s): until cold shutdown is established
Coincident iodine spike	0.025	0.3	

In terms of dose criteria, the limits used in this paper are those from RG 1.195, (NRC, 2003), Table 2, which establish the hypothesis for calculating the dose for persons located at or beyond the boundary of the exclusion area (EAB), low population zone (LPZ) and control room. There is a newer methodology from the NRC, regulated by the RG 1.183, (NRC, 2000) that uses alternative source terms from the RG 1.195 ones for the evaluation of the doses, but it has not been applied in this paper as the PWR-W plants in Spain use the previous methodology.

This paper compares the assumptions and hypotheses made regarding operator action and single failure criteria of those methodologies with and without operator action. This comparison has been made with the same plant model (Almaraz NPP model with NRC's TRACE code), the same best estimate assumptions regarding the initial conditions and the same operator time delays. Therefore, the different single failure criteria is applied to compare the impact on each case.

The objective of this paper is to show the different behaviour of the plant with the different human action and single failure criteria hypothesis used in the safety analysis to see the impact on the calculated offsite dose. Those results are very plant specific, as each plant could have a different single failure criteria and operator action times.

The Almaraz NPP model in TRACE has been developed to be able to simulate the SGTR accident with the precision needed. The simulations covered the considered effective time for offsite dose release, that is, until the last opening of the faulted SG PORV. The offsite dose has been evaluated with the NRC's code RADTRAD, (Humphreys, 1998).

The results revealed significant differences for each human action and single failure criteria assumption in the plant behaviour during the transient, but with similar final results in terms of offsite dose. In addition, some improvements to the Emergency Operating Procedures to lower the offsite release from the damaged SG in case of a SGTR are outlined taking into account the offsite dose sensitive results.

2. Almaraz NPP model for SGTR analysis

Almaraz NPP has two PWR units, it is located in Cáceres (Spain) and it is owned by a consortium of three Spanish utilities: Iberdrola (53%), Endesa (36%) and Gas Natural Fenosa (11%). The commercial operation started in April 1981 (Unit I) and in September 1983 (Unit II). Each unit is a three loop PWR Westinghouse. The nominal power is 2947 MWt and 1055 MWe, respectively. The original Westinghouse steam generators were replaced between 1996 and 1997 and since then it is equipped with three Siemens KWU 61W/D3 steam generators. Reactor coolant pumps are single stage centrifugal model, designed by Westinghouse. The AFWS consists of one turbine driven pump and two motor driven pumps.

Almaraz I NPP TRACE model has 255 thermal–hydraulic components (2 VESSEL, 73 PIPE, 43 TEE, 54 VALVE, 3 PUMP, 12 FILL, 33 BREAK, 32 HEAT STRUCTURE and 3 POWER component), 740 SIGNAL VARIABLES, 1671 CONTROL BLOCKS and 58 TRIPS, Fig. 2.

Regarding the primary and secondary circuits, the following components have been modelled:

- Reactor vessel, modelled by a VESSEL component, Fig. 15, which includes the core region, guide tubes, support columns, core bypass, and the bypass to the vessel head via downcomer and via guide tubes.
- Nuclear core power is modeled with axial and radial cosine power shape distributions. Core power is distributed to nine HEAT STRUCTURE components located each one in one core sector.
- Primary circuit, including steam generators and pressurizer in loop 2 (containing heaters, relief/safety valves and pressurizer spray system).
- Chemical and Volume Control System (CVCS).
- Emergency Core Cooling System (ECCS): safety injection system and accumulators (ACCs).
- The steam lines up to the turbine stop valves, with the relief, safety and isolating valves.
- The steam dump with the eight valves.
- FW and AFW systems. Feed water pumps coastdown and auxiliary mass flows are included as boundary conditions.
- 2D Vessel Pressurizer (PZR) modelling needed for the adequate simulation of the PZR PORV depressurization during the transient.

The control, protection and engineering safeguard systems and signals modelled are:

- Pressurizer level control: CVCS isolating discharge signal, CVCS charge flow and heaters.
- Pressurizer pressure control: including proportional and backup heaters, spray lines and PORVs.
- Steam generators level control system.
- Steam dump control.
- Turbine control.
- Protection and engineering safeguard system-signals: Emergency shutdown system (SCRAM); safety injection; pressurizer safety valve logic; auxiliary feedwater system activation; relief, safety and isolating valve logic of steam lines; normal feedwater system isolation, turbine trip and pump trip.

This model has been validated with steady and transient conditions and verified with a large set of transients, (Queral et al., 2002a), (Queral et al., 2002b), (Queral et al., 2005), (Queral et al., 2008), (Queral et al., 2010), (López et al., 2003) and (González et al., 2005)

For the SGTR simulation, several developments have been accomplished:

- Detailed model of the Steam Dump, needed to simulate the SGTR accident with and without LOOP conditions.
- AFW control detailed modelling, to simulate either the malfunction and the best-estimate operation.
- Several models for the ruptured tube simulation, three of them were selected for performing the calculations, see next section.

2.1. Ruptured tube modelling

Three different options for the ruptured tube modelling were selected, Fig. 3,

- *Model A*: Direct rupture modelling from the pipe representing all the SG U-tubes to the secondary side with a single valve. This option, the most conservative one, is actually used for the most part of the FSAR models.
- *Model B*: The ruptured tube is simulated individually and connected to the secondary side with a valve with double the area of a single tube.
- *Model C*: The ruptured tube is simulated individually and connected to the secondary side with two valves and one in the middle, as it is done normally for double ended breaks in LOCA analysis.

For all the cases it has been proved different chocked flow and pressure drop options, as it is discussed later in the results section.

Model A to C were tested under one tube double ended rupture conditions. The comparison of break mass flow is shown in Fig. 4. The larger break flow has been got with the Model A (direct rupture) and the smaller with Model B (individual tube), being Model

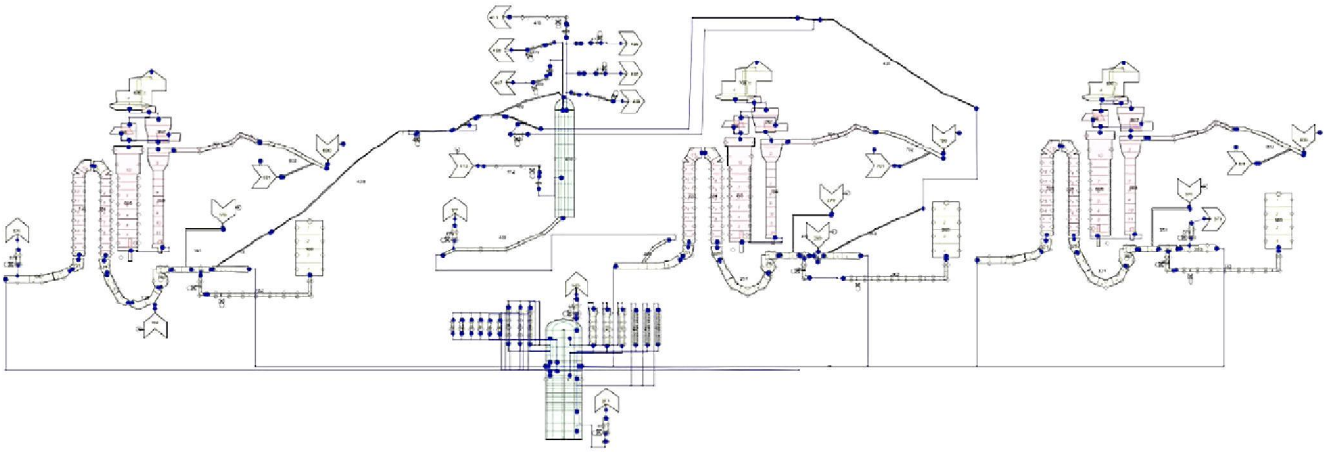

Fig. 2. Simplified scheme of the Almaraz NPP TRACE model.

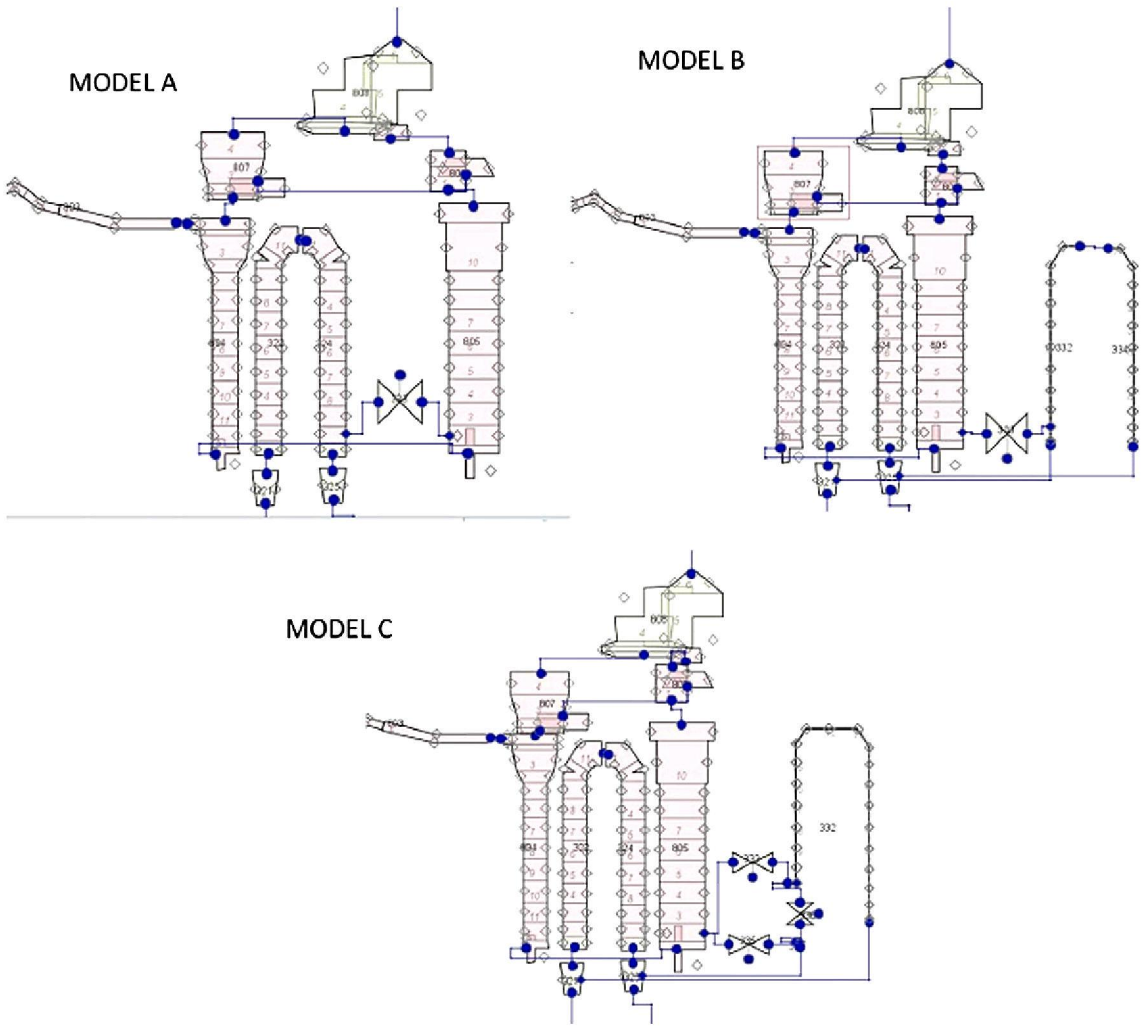

Fig. 3. The three different models chosen for modelling the ruptured tube break.

C (individual tube too) in the middle. The reason for this difference is that: once the individual tube is ruptured, it is depressurized under the pressure of the rest of the primary circuit, having an intermediate pressure between the RCS and the secondary system during the whole transient. In the case of the Model A, as there is not individual ruptured tube model and therefore the pipe representing the whole package of U-tubes is able to maintain the pressure at the same value as the rest of the primary circuit. This difference allows the break flow for model A to be the largest since the pressure drop between the cells connected by the valve representing the rupture is the greatest.

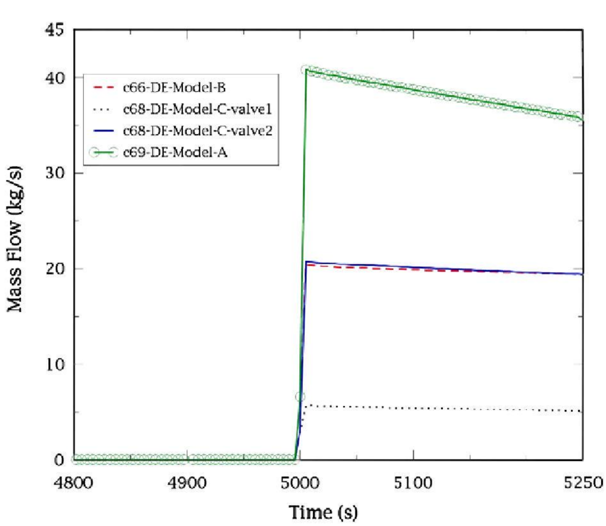

Fig. 4. Mass flow rates for each ruptured tube model.

As it can be seen in Fig. 4, the mass flow of Model B is almost equal to the mass flow of one of the two valves (valve2) of Model C. So, the differences in mass flow are roughly the ones made by the valve1. So, it can be seen that it is not possible to simulate precisely the behaviour of a double ended ruptured tube without modelling it as in Model C, with two valves for the both sides of the rupture and one valve more to avoid mass transfer between the two parts of the tube. This is similar to the way of performing simulation models for LBLOCA analyses.

Taking into account these results it was decided to use the Model C for the calculations (two valves to simulate the doubled ended rupture), as it reproduces more realistically the behaviour of the ruptured tube. Therefore, the Model C is the model used in each one of the SGTR transient simulations of this work.

Compared with the FSAR SGTR simulation cases of the public literature, Table 3, the values for the selected model (around 26 kg/s) are 18% less than the referenced ones, which is the expected result as in the FSAR TH models are more conservative.

Table 3
Aprox. maximum rupture flow rate (kg/s) of several FSAR SGTR simulation analysis.

	Watts Bar	Beaver Valley	Harris NP	Prairie Island	Turkey Point	Braidwood/Byron Stations	Callaway (SNUPPS)
Aprox. maximum rupture flow rate (kg/s)	27.2	39.9	24.5	26.3	38.6	28.6	28.6
Reference document	Monahan and Lewis (1992)	FENOC (2006)	Westinghouse (2000)	Schimmel (2011)	Kiley (2011)	Borton (2011b)	Young (2004)
Figure #	III.7	5.4.2–5	6.3.2–13	2	2	III-3 and III-12	6.4–35
Page #	42	5–339	6.3–46	Enclosure 2, pp. 5	6	III-17 and III-26	6–325

Table 4
Operator times for several real SGTR events and operator studies in full-scope simulators.

Plant/study	Model	SGTR date	Max. leak rate (kg/s)/GPM	At power?	Time operators recognized SGTR (min)	Faulted SG isolation (min)	Reference
Point Beach 1	W-2L	02/26/75	6.0/125	YES	24–28	58	MacDonald et al. (1996)
Surry 2	W-3L	09/15/76	15.9/330	YES	<5	18	MacDonald et al. (1996)
Doel 2	FR-2L	06/25/79	6.5/135	NO	9	9.4	MacDonald et al. (1996)
Prairie Island 1	W-2L	02/10/79	16.2/336	YES	5–18.5	27	MacDonald et al. (1996)
Ginna	W-2L	01/25/82	36.7/760	YES	<1	15	MacDonald et al. (1996)
North Anna 1	W-3L	07/15/87	30.7/637	YES	<5	18	MacDonald et al. (1996)
McGuire 1	W-3L	07/03/89	24.1/500	YES	<1	11	MacDonald et al. (1996)
Mihama 2	W-2L	09/02/91	33.8/700	YES	5	22	MacDonald et al. (1996)
EPRI	W			YES		8.3	EPRI (1989)
KAERI	SIMULATOR TRAINING	1999–2001	Single tube rupture	YES	6.73 (from SCRAM)	19.8 (from SCRAM)	(KAERI, 2005)
HALDEN (Complex /Base scenario)	SIMULATOR TRAINING	2007–2008	Single tube rupture	YES	8–10 min estimated to enter E-3	26.9/16.2	Bye et al. (2011)

In comparison to the maximum mass flow for the real cases, Table 4, the values for the selected model are 15% less than the average of the two double ended ruptures (North Anna and Mihama).

3. Procedures and conditions

To simulate properly the SGTR transient, the following operator actions were implemented:

- Manual isolation of the faulted SG.
- Manual control of AFW.
- Manual depressurization of the secondary system via intact SG PORVs.
- Manual depressurization of the primary system via PZR PORV.
- Manual ending of the SIS.

Those human actions take into account the operator action times. The action times were taken from Watts Bar NPP SGTR FSAR simulation, Table 1 and Reference (Monahan and Lewis, 1992), since they are considered as representative of an FSAR analysis. The reference times for faulted SG isolation from generic Westinghouse plants, (Bochman, 1992), were not considered, since the time for isolation is 10 min, several minutes less than the real cases or the operator studies in full-scope simulators, Table 4.

Once the model were modified and prepared for simulating a SGTR, the different cases were tested. The following assumptions were taken for all cases:

- Cases with and without Loss of Offsite Power coincident with reactor. It means that the RCPs, the PZR spray and the condenser are not available for the accident management.
- The single failure criteria are applied to each one of the methodologies. In the base case with operator actions it was not applied, as an exception.
- The initial conditions are best estimate and base on regular operational data for mid of the cycle.
- For the dose calculation, the assumptions from the RG 1.195 (NRC, 2003) SGTR analysis were taken. The most important assumptions were:
  - Two cases were considered: Coincident Iodine Spike and Pre Accident Spike.
  - The doses were measured for Exclusion Area Boundary (EAB) and Low Population Zone (LPZ). The whole body and thyroid doses were determined for an individual at the most limiting EAB location and for the most limiting receptor at the outer boundary of the LPZ.
  - The dose emission from the faulted SG stops when the affected SG is totally isolated. For the cases with release to the atmosphere via PORVs or Safety Valves, it is assumed that the SG is isolated when the release has finished.
  - All noble gas radionuclides released from the primary system were assumed to be released to the environment without reduction or mitigation.
  - A partition coefficient for iodine of 100 was assumed for the intact SG and 10 for the damaged SG. This has been found to be the most conservative option in the literature and it is used in some FSAR, like the AP1000 one, (Westinghouse, 2007). One of the most common option for the newer methodologies consists on using a partitioning coefficient of 100 for the liquid phase of all the SG and 1.0 for the flashing mass through the break tube.

Regarding the operator action and the single failure criteria, five cases were simulated:

- *Case 1*: The first case simulated was the corresponding with no operator action for the first 30 min, in the same way that the classical FSAR SGTR analysis, (Westinghouse, 1980). Therefore, no operator action was taken credit for the first 1800 s and the break is finished 30 min after the SGTR.

- *Case 2*: The second case was called the *base case* as no single failure criteria was considered, thus it was a reference case of SGTR accident.
- *Case 3*: The third case is the base case but with the faulted SG PORV stuck open at SG isolation and the manual closing of that valve several minutes later as single failure criteria. This is the same single failure criteria that is normally chosen in plants that have applied the WCAP-10698 methodology for offsite dose calculation, (Lewis, et al., 1987b). As the results were very dependent on the SG isolation time, sensitivity analysis was made from 5 min from the tube rupture until 40 min. A crossed sensitivity analysis was done also on the time that the PORV was open, choosing three of the four cases found in the literature:
  - 11 min (Watts Bar NPP, (Monahan and Lewis, 1992),
  - 20 min (Harris NPP, (Westinghouse, 2000) and
  - 30 min (Byron and Braidwood NPPs (Borton, 2011a).
- *Case 4* The fourth case was the base case but with the faulted SG PORV stuck open at SCRAM and the manual closing of that valve several minutes after as single failure criteria. This is the same single failure criteria included in plant analysis like Callaway's FSAR, (Young, 2004). The time that the PORV is closed was chosen to be 20 min after it stuck open, as in Callaway, (Young, 2004). The cases for 11 and 30 min were also run to be compared with the faulted SG PORV stuck open at SG isolation cases.
- *Case 5*: In the last case, a new single failure criteria not analysed before in the public literature for SGTR is proposed. The single failure proposed is the malfunction of the faulted SG level instrumentation during the transient. The malfunction is assumed to start coincident with the tube rupture or after SCRAM, depending on the case. The malfunction consists of giving a fixed value to the faulted SG narrow range level during the transient. This is the value that the control system will use to balance the SG level and also the value that is shown at the control room. Three cases were analyzed:
  - *Case 5A*: the instrumentation is indicating nominal operational narrow range level (about 50%) from SGTR event.
  - *Case 5B*: the instrumentation is indicating zero narrow range level (0%) from SGTR event, that leads to an immediate SCRAM.
  - *Case 5C*: the instrumentation is indicating zero narrow range level (0%) from SCRAM.

4. Results of the SGTR sequences simulations with TRACE 5.0

In this section the main results of the research are shown. The description of the different cases has been included in the previous section. As a summary, the cases presented are:

- *Case 1*: No human action for the first 30 min.
- *Case 2*: Base case: human actions and no single failure criteria.
- *Case 3*: Base case with faulted SG PORV stuck open at SG isolation time as single failure criteria.
- *Case 4*: Base case with faulted SG PORV stuck open at SCRAM as single failure criteria.
- *Case 5*: Base case with faulted SG level instrumentation malfunction as single failure criteria.

The following sensitivities are also analyzed:

- Loss of Offsite Power (all cases). The cases are named as follows:
  - *Case #L*: case with LOOP conditions. Ex: Case 2-L.
  - *Case #NL*: case without LOOP conditions. Ex: Case 2-NL.
- Sensitivity to the damage SG isolation time (Cases 2, 3 and 5A).
- Sensitivity to the damaged SG PORV closing (Cases 3 and 4).
- Sensitivity to the fact of performing or not the cooldown when the RCS temperature is close to the target temperature (Cases 3 and 4).

4.1. Case 1

For this simulation case, there are no manual actions in the first 30 min of the accident. The evolution of the transient in the LOOP case (Case 1-L) is the next one:

- As the SGTR occurs at 5000 s, the pressure begins to drop until 5635 s, when the reactor trip occurs. The Loss of Offsite Power is simultaneous to the SCRAM. Then, there is a quick decrease of the pressure due to the SCRAM.
- Some seconds later, there is a slow increase in the primary pressure due to the SIS action, Fig. 5.
- As there is no manual action to decrease the break flow, it is nearly stable during the first 30 min.
- In the secondary side, all the SG show the same behaviour until the faulted SG isolation occurs at 6800 s.

Since there was no manual depressurization performed, the evolution of the primary and secondary pressures is less abrupt than in a typical SGTR event until until. As a consequence, the integrated mass flow through the ruptured tube is approximately linear, Fig. 6.

The total mass flow through the rupture until the end of the faulted SG PORV opening (30 min after the SGTR) was 35112 kg. The dose amounts are 3.84E−02Sv for the boundary of the exclusion area (EAB) and 1.38E−02Sv for the low population zone (LPZ) in the most limiting case, which was thyroid dose for Coincident Iodine Spike (CIS).

The limit from RG 1.195 for SGTR accident in the CIS case is 0.3Sv, see Table 2 and (NRC, 2003), the results are about 13% (EAB) and 4% (LPZ) of the regulatory limit. The rest of the cases analysed will be compared with this result.

These results are in agreement with the Spanish FSAR for Westinghouse plants, which in the CIS case are in the order of 30% (EAB) and 9% (LPZ) of the thyroid dose limit of RG 1.195, (Queral, 2004). Moreover, the conservatism assumed in the classical FSAR simulation (HPIS mass flow, the rupture modelling, etc.) with a LOFTTR2 model almost doubled the values of a best estimate simulation with a TRACE model.

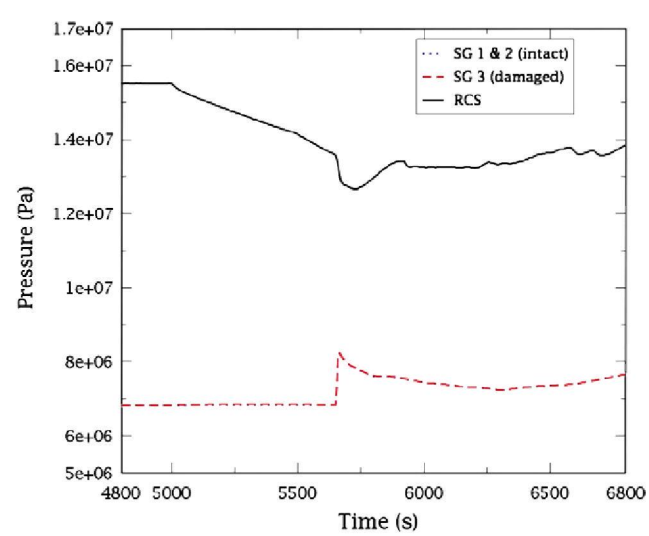

Fig. 5. Case 1-L: RCS and SG pressure.

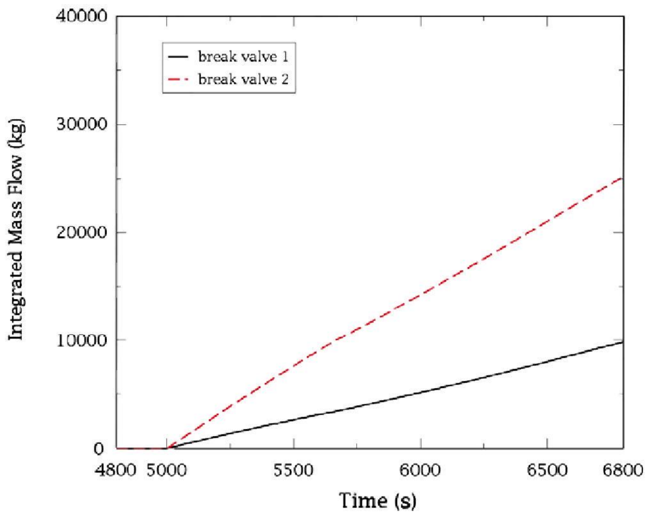

Fig. 6. Case 1-L: Integrated break mass flow.

Regarding the LOOP condition, it can be concluded that is more conservative in terms of EAB thyroid dose, Table 6. This is due to the condenser unavailability the seconds after the SCRAM. In the case with LOOP, the damaged SG PORV is open for a longer time, so that the break mass flow is greater through the ruptured tube. However, although the RCPs action result in a larger mass flow, the fact of having a longer time the damaged SG open makes LOOP condition to be more conservative.

The results are classified in colours depending on the proximity to the Regulatory Guide 1.195 limit, Table 5.

4.2. Case 2

This, called "base case", best estimate assumptions and human actions were included without any single failure criteria. The evolution of the SGTR transient is the next one (Case 2-NL):

- Once the SGTR transient starts, at 5000 s, the RCS pressure starts dropping, Fig. 7.
- Some seconds later, the faulted SG control level lowers the MFW flow entrance to compensate the extra flow that is coming from the ruptured tube.
- In consequence, no increase of level in the ruptured SG is observed after a few seconds of the rupture.
- The pressurizer level starts to decrease and therefore the reactors trips at low pressure level.
- After the SCRAM:
  - All the SG PORVs open immediately due to the pressure increase.
  - The high pressure injection system starts by low pressurizer level.
  - The MFW stops and starts the AFW.
- After 15 min from the SGTR, the faulted SG isolation action is performed.
- Some seconds later, another important human action is performed: the secondary depressurization via the intact SG PORVs. The depressurization starts manually and it is stopped by the operators when the desired temperature in the RCS is reached.
- Then, some minutes later, the primary depressurization is started and it is stopped when the primary circuit has the same pressure as the faulted SG.
- In order to avoid primary repressurization, the SIS is reduced to start equalizing the pressures. With the SIS reduction, the pressurizer level decrease from 90% level to lower values, Fig. 8.

For this base case the rest of actuation until RHR conditions are not simulated as it was out of the scope for this study.

The long term SGTR cases have been analysed in another study by the UPM group with the MAAP code for different sensitivities to the human actions regarding the impact both to core damage and offsite dose consequences, see (Queral, 2012) for the preliminary results.

The results show that the break flow, Fig. 9, behaves different from the no operator action case, at is not continuous and is very dependent on the manual actions being done in the primary and secondary system.

The total discharged mass until the end of the faulted SG PORV opening through the rupture was 31,592 kg. The most limiting dose calculated was thyroid dose in EAB for coincident iodine spiking, and its value was 90% of the case with no human action, which corresponds to 12% of the RG 1.195 limit.

To see the impact of the LOOP and the HPIS mass flow, two extra cases were analysed:

- Base case with LOOP (Case 2-L).
- Base case without LOOP but with only one HPIS train available (Case 2-L-1T).

The case with LOOP results in more conservative, as the total mass flow until the end of the faulted SG PORV opening through the rupture was 34,991 kg, 10% more than the base case without LOOP. The case with one train of HPIS is less conservative, as the

Table 5
Color code for the classification of the results.

	0-20	20-40	40-60	60-100
Compared to the RG 1.195 dose limit (%)	(green)	(yellow)	(orange)	(red)

Table 6
Results of Case 1 in terms of most limiting dose value obtained (Thyroid dose in EAB for CIS).

Simulation case	EAB thyroid dose obtained (2h) Compared to the Case 1-L | Compared to the RG 1.195 dose limit (%)			
	Case 1-L (with LOOP)		Case 1-NL (No LOOP)	
Case 1	1.00	12.8%	0.96	12.3%

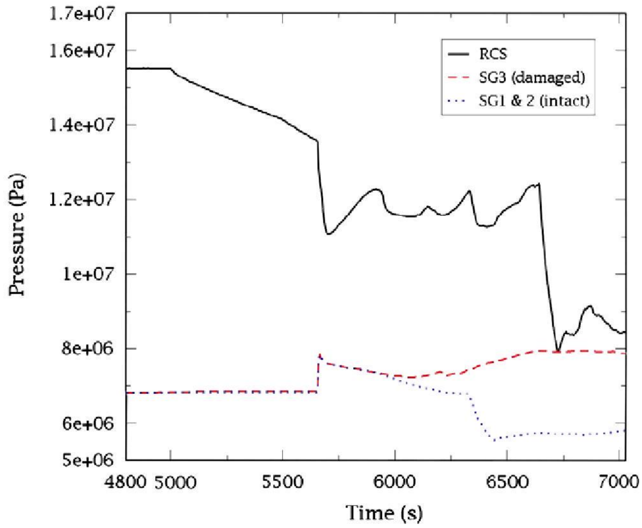

Fig. 7. Case 2-NL: RCS and SG pressure.

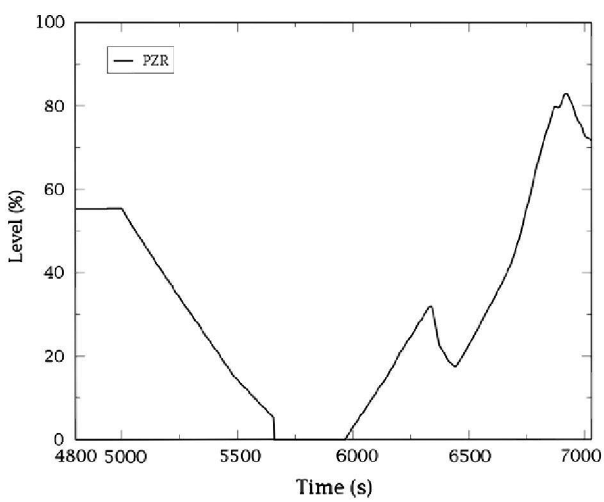

Fig. 8. Case 2-NL: Pressurizer level.

integral break mass flow is 7% less than in the base case with 2 HPIS trains available.

From Case 2 global results, Table 7, it is confirmed that for best estimate conditions, the LOOP assumption is more conservative in

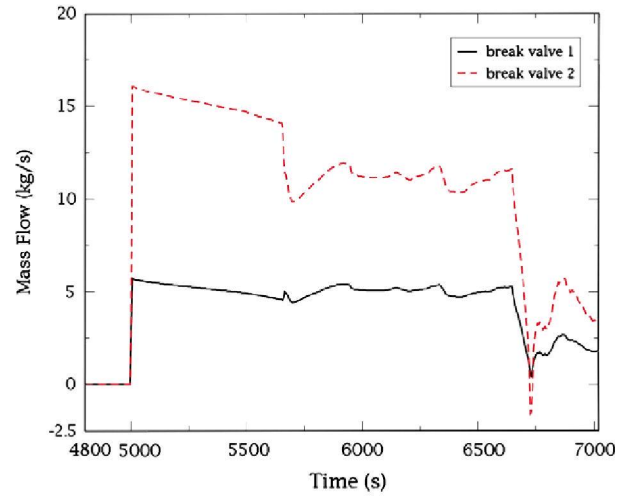

Fig. 9. Case 2-NL: Break mass flow in both sides of the ruptured tube.

terms of off-site dose,although the difference is less than for the Case 1.

Another effect is captured: the fact of having 2 HPIS trains (the normal operating mode) in an SGTR is more conservative in terms of dose that having one HPIS train failed and one HPIS train working. This issue has inspired a proposal of change to the Emergency Operating Procedures, as it will be detailed in the next section. It seems from the Case 2 results that assuming operator actions from the beginning of the transient with best estimate assumptions (no single failure criteria) has a similar result in terms of limiting dose as the case with no operator action for the first 30 min.

It needs to be remember that in Case 1 although is assumed that no operator action is made in the first 30 min, the SG is isolated in minute number 30, therefore the dose contribution of the break mass flow stops there. In the case with operator action (Case 2), the faulted SG has a contribution to the dose for more than 30 min after the tube rupture. With the operator inaction the break flow is greater that assuming operator action from the beginning, so the combination of both arguments make the integral mass flow though the rupture at the SG isolation time to be a similar quantity.

4.3. Case 3

The simulation of this case is more complex, as it involves more actuations of the operating crew during the transient than in the base case (Case 2).

Table 7
Results of Case 2 in terms of most limiting dose value obtained (Thyroid dose in EAB for CIS).

Simulation case	EAB thyroid dose obtained (2h) Compared to the Case 1-L | Compared to the RG 1.195 dose limit (%)											
Case 2	Case 2-L (with LOOP)						Case 2-NL (No LOOP)					
							2 HPIS trains			1 HPIS train		
	1.05			13.4%			1.02		13.4%	0.95		12.2%
Isolation action time	15 min		20 min		25 min		30 min		35 min		40 min	
Case 2-L	1.05	13.4%	1.48	19.0 %	2.06	26.4%	2.61	33.4%	3.32	42.5%	4.04	51.8%

The behaviours of the pressure in the RCS and the faulted SG are quite different to the Case 2, as it can be seen in Fig. 10. The evolution of the SGTR transient is the same than the Case 2 until simultaneously to the SG isolation; the faulted SG PORV failed open for 11 min. Then, the sequence of events is different to Case 2:

- The pressure at the faulted SG starts to decrease suddenly due to the PORV opening, cooling and depressurizing the primary circuit too.
- After the faulted SG PORV isolation, when it is supposed to start the secondary depressurization, the primary temperature is very low and approaching to the target temperature for the secondary depressurization (about 20 K below the saturation temperature of the faulted SG), Fig. 11.
- Therefore, it is assumed that the secondary side depressurization is worth and it is started in second 6900.
- Later on, the primary depressurization via PZR PORV is started until the RCS and the faulted SG has the same pressure.
- Seconds later the SIS is stopped in order to avoid an increase in the RCS pressure again.

As mentioned in the beginning of the section, it has been analyzed an extra case without RCS cooldown, to measure the impact on the transient evolution. In this case, it is assumed that is no need to start the RCS cooldown as the primary is cold enough to start the PZR PORV depressurization:

- It starts about second 6900, depressurizing the primary circuit until the faulted SG pressure.
- Seconds later, the SIS is stopped in order of not to increase the pressure again.

The difference between the cases with and without RCS cooldown is that without cooldown it is not possible to stop the leakage without a second RCS depressurization. It can be concluded that the secondary depressurization for this case worth because leads the RCS to a more stable condition that can avoid a second PZR depressurization.

The total mass discharged until the end of the faulted SG PORV opening through the rupture is 35,112 kg in both cases. The most limiting dose calculated was thyroid dose in EAB for coincident io-

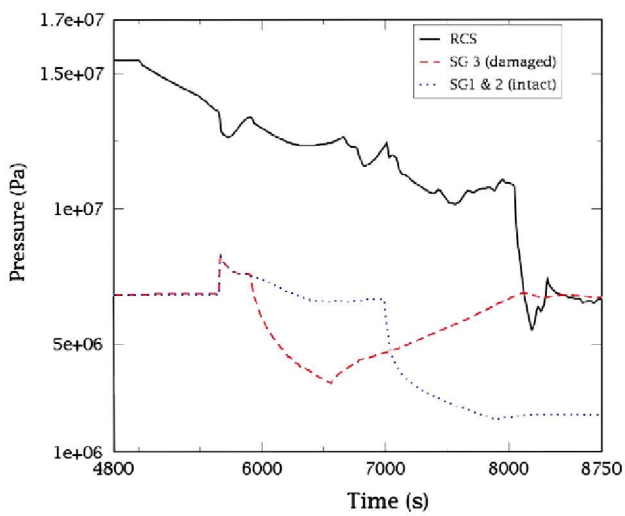

Fig. 10. Case 3-L: RCS and SG pressure. Case with secondary depressurization.

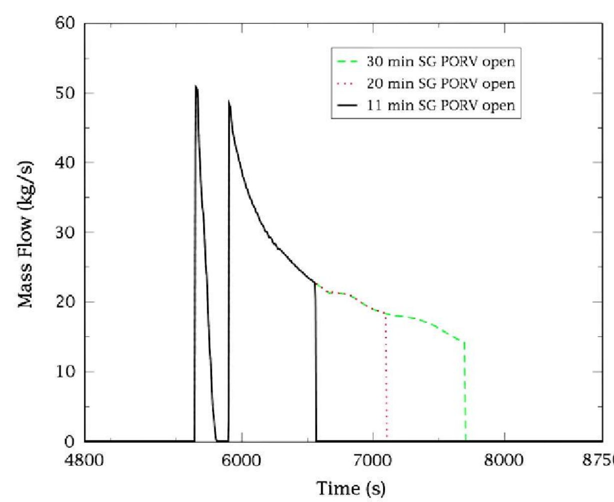

Fig. 12. Case 3-L: Sensitivity to SG-PORV opening time. Faulted SG PORV mass flow.

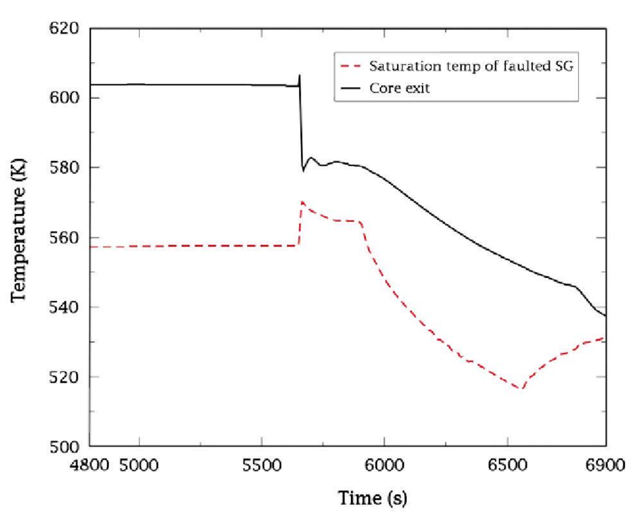

Fig. 11. Case 3-L: Core exit temperature vs. faulted SG saturation temperature before depressurizing the secondary system.

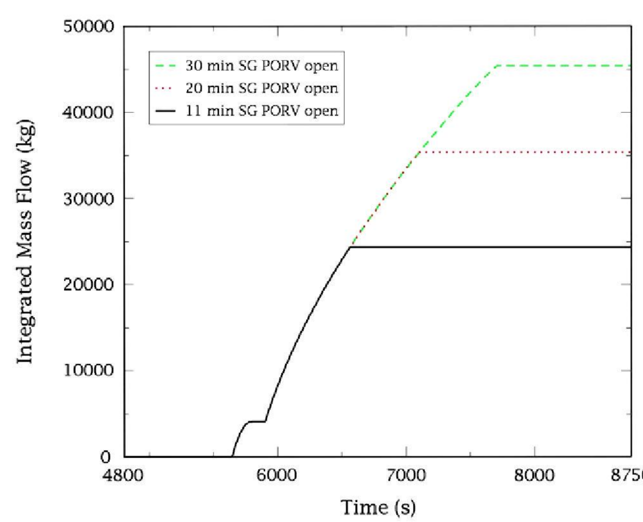

Fig. 13. Case 3-L: Sensitivity to SG-PORV opening time. Faulted SG PORV integrated mass flow.

Table 8
Results of Case 3 in terms of most limiting dose value obtained (Thyroid dose in EAB for CIS).

Simulation case	EAB thyroid dose obtained (2h) Compared to the Case 1-L | Compared to the RG 1.195 dose limit (%)			
Case 3	Case 3-L (with LOOP)		Case 3-NL (No LOOP)	
	0.78	9.93%	0.74	9.43%

Simulation case	EAB thyroid dose obtained (2h) Compared to the Case 1-L | Compared to the RG 1.195 dose limit (%)															
	Case 3-L															
Isolation /PORV stuck open time	5 min		10 min		15 min		20 min		25 min		30 min		35 min		40 min	
30 min	1.30	16.6%	1.67	21.4%	2.00	25.6%	2.39	30.6%	3.14	40.2%	4.03	51.7%	4.82	61.8%	5.72	73.4%
20 min	0.81	10.4%	1.08	13.9%	1.35	17.3%	1.73	22.1%	2.24	28.6%	2.99	38.3%	3.74	47.9%	4.41	56.6%
11 min	0.32	4.2%	0.52	6.6%	0.78	9.9%	1.12	14.4%	1.51	19.3%	1.98	25.3%	2.52	32.3%	3.08	39.4%

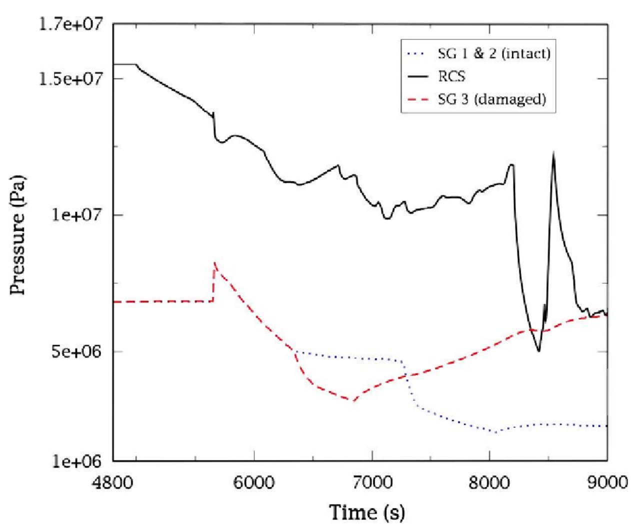

Fig. 14. Case 4-L: RCS and SG pressure.

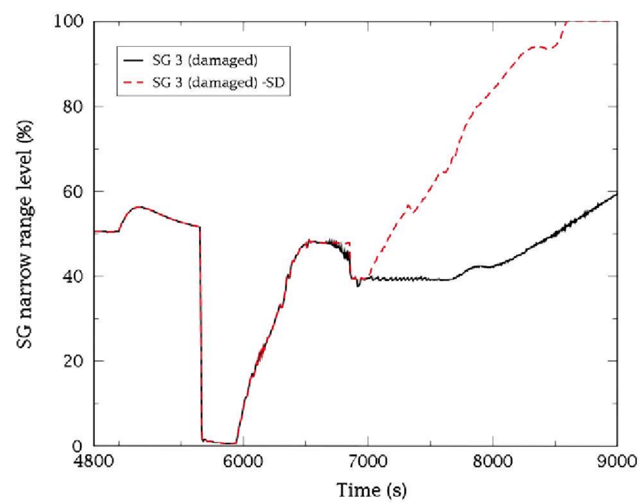

Fig. 15. Case 4-L: Faulted SG narrow range level. Cases with and without secondary side depressurization (SD case means with secondary depressurization).

Table 9
Results of Case 4 in terms of most limiting dose value obtained (Thyroid dose in EAB for CIS).

Simulation case	EAB thyroid dose obtained (2h) Compare to the Case 1-L | Compare to the RG 1.195 dose limit (%)			
Case 4	Case 4-L (with LOOP)		Case 4-NL (No LOOP)	
	1.02	13.0%	0.78	10.0%
PORV stuck open time	Case 4-L			
30 min	1.94		24.8%	
20 min	1.02		13.0%	
11 min	0.55		7.0%	

Table 10
Results of Case 5 in terms of most limiting dose value obtained (Thyroid dose in EAB for CIS).

Simulation case	EAB thyroid dose obtained (2h) Compare to the Case 1-L | Compare to the RG 1.195 dose limit (%)											
Case 5	Case 5A-L						Case 5B-L					
	1.15			14.8%			0.93			11.9%		
Case 5A	Case 5A-L (with LOOP)						Case 5A-NL (No LOOP)					
	1.15			14.8%			3.63			46.6%		
Isolation	15 min		20 min		25 min		30 min		35 min		40 min	
Case 5-L	1.15	14.8%	1.72	22.0%	2.27	29.1%	2.87	36.8%	3.55	45.5%	4.23	54.2%

dine spiking, and its value was 91% of the case with no human action, which is 12% of the RG 1.195 limit.

As a first time sensitivity analysis, three cases were done with the fixed SG isolation time at 15 min after the SGTR with faulted SG PORV opening during 11, 20 and 30 min, as it was pointed previously. The faulted SG PORV release, Figs. 11–13, is clearly dependent on the period of time that the valve remains open, The total dose is close related with this time, as the radioactive material is released until the valve is closed.

Therefore, there has been done a second time sensitivity analysis including times of SG isolation from 5 min from the start of the SGTR to 40 min, in steps of 5 min. Each one of these cases has been simulated also for faulted SG PORV opening of 11, 20 and 30 min.

The results of both sensitivity analysis for Case 3, Table 8, show clearly that the final limiting dose is more sensitive to the time that the PORV remains open than the time of SG isolation.

For the most limiting cases, the results are more restrictive than the no operator action case (Case 1) or the base case (Case 2), but it is necessary to say that the times chosen are quite improbable looking at the different hypothesis made by the plants in their simulations. For more probable isolation times, like 15 or 20 min, the dose results are from 0.80 to 2.4 times the no operator action case (from 10% to 31% of the RG 1.195 limit).

Finally, it is clear also that the LOOP case is again more conservative, but as for Case 2 the differences are smaller than for Case 1.

4.4. Case 4

In this case, the behaviour of the plant is quite different from the previous three cases in the first phase:

- As the faulted SG PORV fails open before its isolation, at reactor trip, all the SG are depressurized at the same time for 20 min, Fig. 14.
- As it happened in the previous case with SG PORV stuck open, it is assumed again that that the secondary depressurization is worth and it is started in second 7270.
- Later on, the primary depressurization via PZR PORV is started until the RCS and the faulted SG has the same pressure.
- The pressure suddenly increases until the SIS is stopped.

It is not possible to stop the leakage without a second RCS depressurization in the short term, although the pressures are stabilized. An important fact is that the faulted SG is filled up with water before the primary depressurization starts, Fig. 15, although

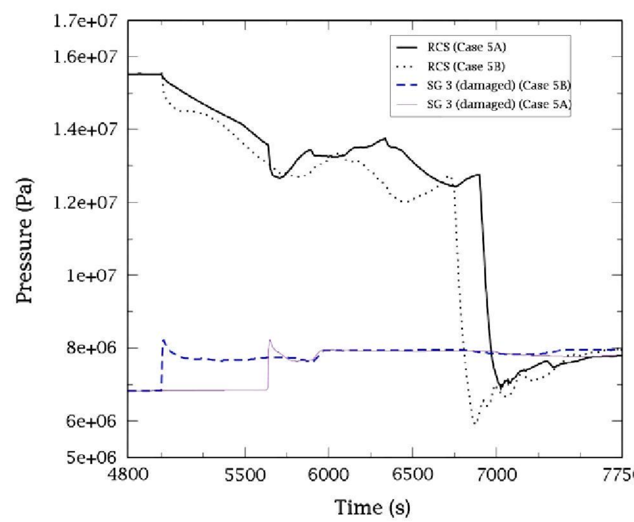

Fig. 16. Cases 5A-L and 5B-L: RCS and SG pressure.

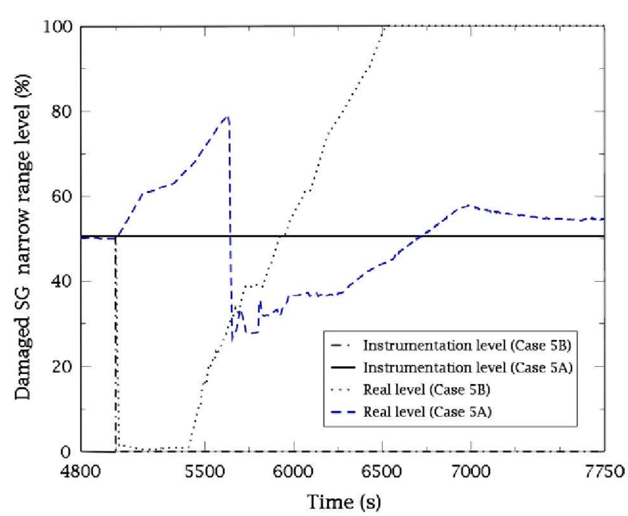

Fig. 17. Cases 5A-L and 5B-L: Damaged SG instrumentation and real level.

no water release through the PORV has been detected during the simulation.

As in Case 3, a case without cooldown is performed. As in the case with cooldown, it is not possible to stop the leakage without a second primary depressurization in the short term, although the pressures are stabilized. One important fact in this case is that as the leakage lasts less than in the case with the cooldown, so the faulted SG is not filled with water. It can be concluded that for this case it is better not to start the cooldown if the RCS temperature is close to the target and it is more appropriate to start the PZR depressurization directly.

The total discharged mass until the end of the faulted SG PORV opening through the rupture is 35,254 kg. The most limiting dose calculated was thyroid dose in EAB for coincident iodine spiking, and its value was 101% of the case with no human action, which is 13% of the RG 1.195 limit.

The results of Case 4, Table 9, show that although the behaviour of the plant is quite different than in the Case 3. In terms of limiting dose the results are quite similar but lower than the case with the same time of PORV opening (20 min) with SG isolation at 10 min, so it can be concluded that is more limiting to have the faulted SG PORV stuck open after SG isolation than having it at SCRAM.

From the sensitivity analysis of Case 3 and Case 4, it can be concluded that both increasing the SG isolation time or the time the faulted SG PORV is open the off-site dose is increased greatly. Therefore, it is needed both to minimize the time of SG isolation and the time that the PORV stucks open. This fact has inspired the other EOP proposal, detailed in the next chapter.

Fig. 18. Management of the damaged SG PORV stuck open at damaged SG MSIV closing with EOP E-3 and E-2.

Finally, it is clear also that the LOOP case is again more conservative, but in this case the difference is greater than for Cases 2 and 3. The explanation for this difference is that the LOOP condition affects more if the PORV opening is closer to the SCRAM, as it happens in Case 4 compared to Case 3.

4.5. Case 5

In this section, the results of the simulation of the SGTR under proposed new single failure criteria are showed. The single failure proposed is the malfunction of the faulted SG level instrumentation during the transient. The malfunction is assumed to start coincident with the tube rupture or after SCRAM, depending on the case.

The malfunction consists of giving a fixed value to the faulted SG narrow range level during the transient. This is the value that the control system will use to balance the SG level and also the value that is shown at the control room. Three cases were analyzed:

- *Case 5A*: The instrumentation is indicating nominal operational narrow range level (about 50%) from SGTR event.
- *Case 5B*: The instrumentation is indicating zero narrow range level (0%) from SGTR event.
- *Case 5C*: The instrumentation is indicating zero narrow range level (0%) from SCRAM.

In the *Case 5A*:

- The main feedwater system does not lower the flow when the SGTR starts and the damaged SG level increase before the SCRAM.
- The reactor is depressurized at second 6300 by the intact SGs and later on by the PZR PORV, at second 7000.
- This second depressurization is very effective that when it is stopped, the primary pressure is below the faulted SG one.
- As a consequence, the PZR becomes solid as the flow go inverse. This sequence can lead to a boron dilution scenario, but it is expected to be small in magnitude.
- When the SIS is stopped, the pressures are equalized and the break flow is finished.

The total mass flow until the end of the faulted SG PORV opening through the rupture is 36,712 kg. The most limiting dose calculated was thyroid dose in EAB for coincident iodine spiking, and its value was 105% of the case with no human action, which is 13% of the RG 1.195 limit.

From Case 5A results, Table 10, the LOOP case is less conservative. In this case the SCRAM was coincident with the SGTR, therefore the LOOP conditions are not so relevant. For Case 5A it is detected the same trend to increase the off-site dose with the time SG isolation time.

In the *Case 5B*, the reactor SCRAM is instantaneous when the damaged SG instrumentation level drops to 0% in the narrow range at SGTR event. The behaviour of the primary and secondary pressures is quite similar to the 5A case, Fig. 16. The important difference with 5A case is that the damaged SG is filled very quickly, less than 25 min, before cooldown and RCS depressurization, Fig. 17.

Nevertheless no liquid mass flow to the atmosphere is released after this fact. It happens because the damaged SG secondary pressure is quite low compared with the PORV and RV setpoint.

The Case 5B is less conservative than the Case 5A at the beginning of the transient until the SCRAM occurs, as the level of the faulted SG rises quick, but then is more conservative, as the feed from the AFW is less than in the other cases because the level is artificially higher, as the instrumentation still gives the nominal operating value of 50%, being the real level less than that for some minutes.

The Case 5B is less limiting than the Case 5A in terms of offsite dose, as the damaged SG is being filled with AFW water, slowing the break flow. Nevertheless in 5B cases the damaged SG is filled up, but without any water release through the PORV.

The *Case 5C* is similar to the case 5B, but gets an extra mass flow from until the SCRAM. The results are quite close to the 5B case, so they will not be detailed.

5. Proposal of change for E-3 Emergency Operating Procedure to minimize the offsite dose in case of a SGTR sequence

In this section, two proposal of change for EOP E-3 are outlined, regarding the results obtained in the simulations. The purpose of these proposals is to minimize the offsite dose during the SGTR sequence management. It applies to all the PWR of Westinghouse design.

5.1. First proposal: HPIS reduction

BACKGROUND: It has been analysed the Case 2 with one and two train of HPIS available. Some facts must be outlined:

- The cooldown and depressurization actions are quite more effective in the one HPIS train case than in the 2 HPIS trains case. This effect is possible due to the pressurizing effect of the SIS.
- The offsite dose is almost 6% less in the one HPIS train case than for the 2 HPIS trains case.
- In any of both cases there is risk to uncover the core, as one HPIS train is more than enough to compensate the break mass flow through the tube.

PROPOSAL: It may be necessary to study the action of reducing the HPIS to one train if a SGTR is detected and the pressure drop is less than the equivalent of 1 tube double ended guillotine ruptured.

BENEFITS: This change in the E-3 EOP might allow lowering the offsite release and to manage the cooldown and depressurization much more effectively.

5.2. Second proposal: faulted SG PORV status as continuous action step

BACKGROUND: the fact of having a faulted SG PORV stuck open after MSIV closing is quite sensitive from an operational point of view:

- The MSIV is closed in the sub step 3h of E-3 EOP and the faulted SG PORV has been checked in sub step 3b, Fig. 18.
- If the operator does not want to violate the procedure, he must continue with the E-3 procedure until step 6 and then to step 1 of E-2 “Faulted SG Isolation” EOP.
- Starting the E-2 procedure, he has to go through it until the PORV is told to be locally isolated in Step 4.
- In the Step 6, as there is secondary radiation alarm, it drives the operation crew again to E-3 EOP, but to Step 1 again.
- Conclusions:
  - The fact of not violating the procedure and wait to close locally the damaged SG PORV can drive to a delay of more than 10 min.
  - Having the damaged SG PORV 10 additional minutes open represents 20% more of offsite dose, as it is can be seen in Table 8.

PROPOSAL: checking the damaged SG PORV status and closing it (locally if needed) is recommended to be upgrade to CONTINOUS ACTION STEP in E-3 EOP, *BENEFITS*: it will avoid unnecessary additional offsite dose in case of a SGTR with faulted SG PORV stuck open.

6. Conclusions

In this paper, several human action and single failure criteria for SGTR analysis have been compared with a model of Almaraz NPP using a best estimate code as TRACE 5.0.

Until the global conclusion about the single failure criteria is that, in case of not having any single failure criteria, the classical FSAR assumption of no operator action for the first 30 min (Case 1) is the most conservative. Nevertheless, the results are similar to the best-estimate case with human actions (Case 2). This fact is really interesting, as it confirms the idea that modelling the operator action with realistic operator action times is equivalent in terms of offsite dose to no operator action in the first 30 min, which was the key to build the classical FSAR methodology, (Westinghouse, 2009).

In the case of considering a single failure criteria (Cases 3–5), unless it is assumed that the operating crew is able to isolate the SG in less than 15 min, which is not conservative at all regarding the real cases, the offsite dose normally is greater than in the no operator action case (Case 1).

The time that takes the operating crew to isolate the SG need to be carefully determined through operating crew training, as the real cases show values higher than the chosen ones for the FSAR analysis, as it can be seen in the previous sections. The other important parameter, the time needed for closing the faulted SG PORV once is stuck open is also critical and needs to be determined by simulation/training on-site. This time it is very plant specific, as the physical layout of the plant affects very much to the time spent to reach the SG PORV and close it by the plant personnel.

Some improvements to the EOPs to lower the offsite dose are suggested: to upgrade the damaged SG PORV status to a CONTINOUS ACTION STEP in EOP E-3 and to study the adequacy of reducing the HPIS to one train in case of a mayor leak or single tube SGTR in order to avoid unnecessary radioactive release and to manage more adequately the cooldown and depressurization.

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